Duke Energy

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September 30, 2002

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Duke Energy Corporation
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Proposed Technical Specifications and Bases Amendment
3.6.16.1 and 3.6.16.3; and Administrative Controls 5.5.2
Reactor Building Integrity

Pursuant to 10CFR50.90, this letter submits a license amendment request (LAR) for the McGuire Nuclear Station Facility Operating Licenses and Technical Specifications (TS). This amendment applies to Surveillance Requirements (SR) 3.6.16.1 and 3.6.16.3 for Reactor Building Integrity, and Administrative Control 5.5.2 concerning the Containment Leakage Rate Testing Program. The changes proposed by Duke Energy Corporation (Duke) in this amendment request will: 1) modify the details of the Surveillance Requirement to be consistent with the design of the Reactor Building access openings[1], 2) modify the frequency of the Surveillance Requirement for visual inspections for the exposed interior and exterior surface of the reactor building, and 3) modify the administrative controls for the Containment Leakage Rate Testing Program.

[1] The reactor building is a concrete structure that surrounds the steel containment building. The access doors to the reactor building referred to herein differ from those of the containment airlock, through which access is gained to the containment building.

A017

Please note that McGuire and Catawba proposed a license amendment request on May 29, 2002 (Containment Leakage Rate Testing Program) which also affects a TS page enclosed in this license amendment request. Therefore, the two license amendment requests will need to be coordinated regarding the processing and approval of changes to this page. Duke Energy Corporation will submit any necessary revised TS pages resulting from NRC approval of this previous license amendment request.

The contents of this amendment package are as follows:

Attachment 1 provides marked copies of the affected TS and Bases pages for McGuire showing the proposed changes.

Attachment 2 contains reprinted pages of the affected TS and Bases pages for McGuire.

Attachment 3 provides a description of the proposed changes and technical justification.

Pursuant to 10CFR50.92, Attachment 4 contains the results of the No Significant Hazards determination.

Pursuant to 10CFR51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from the performance of an Environmental Assessment/Impact Review.

The proposed amendment is similar to that submitted by the Catawba Nuclear Station on February 18, 1999, and approved by the NRC in the Safety Evaluation Report (SER) dated April 9, 1999.

Implementation of this amendment will not impact the McGuire Updated Final Safety Analysis Report (UFSAR).

NRC approval of this License Amendment Request is requested by August 2003 in order to support the next McGuire Unit 2 reactor building inspection.

In accordance with Duke administrative procedures and Quality Assurance Program Topical Report requirements, this proposed amendment has previously been reviewed and approved by the McGuire Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10CFR50.91, a copy of this proposed amendment is being sent to the appropriate state official.

Inquiries on this matter should be directed to J. A. Effinger at (704) 382-8688.

Very truly yours,

HB Barron

H. B. Barron

H. B. Barron, being duly sworn, affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

HB Barron

H. B. Barron, Vice President

Subscribed and sworn to me: 9-30-02
Date

Deborah S. Rome, Notary Public
Deborah S. Rome

My commission expires: December 19, 2004

SEAL

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ATTACHMENT 1

McGUIRE UNITS 1 AND 2 TECHNICAL SPECIFICATIONS
AND
TECHNICAL SPECIFICATION BASES

MARKED COPY

3.6 CONTAINMENT SYSTEMS

3.6.16 Reactor Building

LCO 3.6.16 The reactor building shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.16.1	Verify ~~each~~ THE door in each access opening is closed, except when the access opening is being used for normal transit entry and exit~~, then, at least one door shall be closed.~~	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.16.2	Verify each Annulus Ventilation System train produces a pressure equal to or more negative than -0.5 inch water gauge in the annulus within 22 seconds after a start signal and -3.5 inches water gauge after 48 seconds. Verifying that upon reaching a negative pressure of -3.5 inches water gauge in the annulus, the system switches into its recirculation mode of operation and that the time required for the annulus pressure to increase to -0.5 inch water gauge is ≥ 278 seconds.	18 months on a STAGGERED TEST BASIS
SR 3.6.16.3	Verify reactor building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the reactor building.	~~40 months~~ ~~AND~~ ~~During shutdown for SR 3.6.1.1 Type A tests~~ 3 TIMES EVERY 10 YEARS, COINCIDING WITH CONTAINMENT VISUAL EXAMINATIONS REQUIRED BY SR 3.6.1.1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.16 Reactor Building

BASES

BACKGROUND

The reactor building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the reactor building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the reactor building and the steel containment vessel under post accident conditions. Filters in the system then control the release of radioactive contaminants to the environment. The reactor building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS. To ensure the retention of containment leakage within the reactor building:

a. THE ~~Each~~ door in each access opening is closed except when the access opening is being used for normal transit entry and exit~~, then at least one door shall be closed~~ and

b. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The design basis for reactor building OPERABILITY is a LOCA. Maintaining reactor building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The reactor building satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Reactor building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

BASES

APPLICABILITY

Maintaining reactor building OPERABILITY prevents leakage of radioactive material from the reactor building. Radioactive material may enter the reactor building from the containment following a LOCA. Therefore, reactor building OPERABILITY is required in MODES 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, reactor building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event reactor building OPERABILITY is not maintained, reactor building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

B.1 and B.2

If the reactor building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.16.1

Maintaining reactor building OPERABILITY requires maintaining ~~each~~ THE door in ~~the~~ EACH access opening closed, except when the access opening is being used for normal transit entry and exit ~~then at least one door must remain closed~~. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available ~~to the operator~~.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.16.2

The ability of a AVS train to produce the required negative pressure within the required times provides assurance that the building is adequately sealed. The negative pressure prevents leakage from the building, since outside air will be drawn in by the low pressure. The negative pressure must be established within the time limit to ensure that no significant quantity of radioactive material leaks from the reactor building prior to developing the negative pressure.

The AVS trains are tested every 18 months on a STAGGERED TEST BASIS to ensure that in addition to the requirements of LCO 3.6.10, "Annulus Ventilation System," either AVS train will perform this test. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage.

SR 3.6.16.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the reactor building. The ~~40 month~~ Frequency is based on ~~the requirement to perform two additional inspections at approximately equal intervals between the Type A tests required by SR 3.6.1.1 performed on a 10 year interval. The verification is done during shutdown.~~

ENGINEERING JUDGEMENT, AND IS THE SAME AS THAT FOR CONTAINMENT VISUAL INSPECTIONS PERFORMED IN ACCORDANCE WITH SR 3.6.1.1.

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Station Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995~~.~~ EXCEPT THAT THE CONTAINMENT VISUAL EXAMINATIONS REQUIRED BY REGULATORY POSITION C.3 SHALL BE CONDUCTED 3 TIMES EVERY 10 YEARS, INCLUDING DURING EACH SHUTDOWN FOR SR 3.6.1.1 TYPE A TESTS, PRIOR TO INITIATING THE TYPE A TEST.

(continued)

ATTACHMENT 2

McGUIRE UNITS 1 AND 2 TECHNICAL SPECIFICATIONS
AND
TECHNICAL SPECIFICATION BASES

REPRINTED VERSION

Remove Page	Insert Page
3.6.16-1	3.6.16-1
3.6.16-2	3.6.16-2
B3.6.16-1	B3.6.16-1
B3.6.16-2	B3.6.16-2
B3.6.16-3	B3.6.16-3
5.5-1	5.5-1

3.6 CONTAINMENT SYSTEMS

3.6.16 Reactor Building

LCO 3.6.16 The reactor building shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.16.1 Verify the door in each access opening is closed, except when the access opening is being used for normal transit entry and exit.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.16.2	Verify each Annulus Ventilation System train produces a pressure equal to or more negative than -0.5 inch water gauge in the annulus within 22 seconds after a start signal and -3.5 inches water gauge after 48 seconds. Verifying that upon reaching a negative pressure of -3.5 inches water gauge in the annulus, the system switches into its recirculation mode of operation and that the time required for the annulus pressure to increase to -0.5 inch water gauge is ≥ 278 seconds.	18 months on a STAGGERED TEST BASIS
SR 3.6.16.3	Verify reactor building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the reactor building.	3 times every 10 years, coinciding with containment visual examinations required by SR 3.6.1.1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.16 Reactor Building

BASES

BACKGROUND

The reactor building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the reactor building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the reactor building and the steel containment vessel under post accident conditions. Filters in the system then control the release of radioactive contaminants to the environment. The reactor building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS. To ensure the retention of containment leakage within the reactor building:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The design basis for reactor building OPERABILITY is a LOCA. Maintaining reactor building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The reactor building satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Reactor building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

BASES

APPLICABILITY

Maintaining reactor building OPERABILITY prevents leakage of radioactive material from the reactor building. Radioactive material may enter the reactor building from the containment following a LOCA. Therefore, reactor building OPERABILITY is required in MODES 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, reactor building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event reactor building OPERABILITY is not maintained, reactor building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

B.1 and B.2

If the reactor building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.16.1

Maintaining reactor building OPERABILITY requires maintaining the door in each access opening closed, except when the access opening is being used for normal transit entry and exit. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.16.2

The ability of a AVS train to produce the required negative pressure within the required times provides assurance that the building is adequately sealed. The negative pressure prevents leakage from the building, since outside air will be drawn in by the low pressure. The negative pressure must be established within the time limit to ensure that no significant quantity of radioactive material leaks from the reactor building prior to developing the negative pressure.

The AVS trains are tested every 18 months on a STAGGERED TEST BASIS to ensure that in addition to the requirements of LCO 3.6.10, "Annulus Ventilation System," either AVS train will perform this test. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage.

SR 3.6.16.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the reactor building. The Frequency is based on engineering judgment and is the same as that for containment visual inspections performed in accordance with SR 3.6.1.1.

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Station Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, except that the containment visual examinations required by regulatory position c.3 shall be conducted 3 times every 10 years, including during each shutdown for SR 3.6.1.1 Type A tests, prior to initiating the Type A test.

(continued)

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATON

Proposed Change 1

Currently, Surveillance Requirement 3.6.16.1 of the McGuire Technical Specifications states the following:

"Verify each door in each access opening is closed, except when the access opening is being used for normal transit entry and exit; then at least one door shall be closed." McGuire Technical Specifications require this verification to be conducted on a 31-day frequency.

McGuire proposes to amend this Surveillance Requirement to state:

"Verify the door in each access opening is closed, except when the access opening is being used for normal transit entry and exit."

Justification for the proposed change is as follows:

NUREG-1431, Revision 1, "Standard Technical Specifications, Westinghouse Plants," (the version from which McGuire's Improved Technical Specifications were prepared) contained a version of this Surveillance Requirement which stated, "Verify each door in each access opening is closed, except when the access opening is being used for normal transient entry and exit [; then, at least one door shall be closed]." (Please note that McGuire changed the editorially incorrect "transient" of NUREG-1431 to the editorially correct "transit" in the McGuire plant specific Technical Specifications.

The information in brackets was intended to be included in the plant specific version of the Surveillance Requirement only if it was applicable to the facility. The information in brackets applied to plants which utilize an airlock design (two doors) for entry into the reactor building. McGuire has five (5) openings into the reactor building. These are: the fuel building door to the upper airlock, the auxiliary building door to the upper airlock, the auxiliary building door to the upper annulus, the auxiliary building door to the lower airlock, and the auxiliary building door to the lower annulus. Each of these openings has only a single door.

The information contained in brackets should not have been included in the Technical Specifications because it was not applicable to McGuire. The information was included because it was erroneously included in the previous version of the Technical Specifications (i.e., prior to conversion to the Improved Technical Specifications). This information was included in the previous McGuire definition of REACTOR BUILDING INTEGRITY which stated, in part:

"REACTOR BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed, …"

Since the incorrect information in the previous version of the Technical Specifications formed the basis for the appearance of this same information in the Improved Technical Specifications, deletion of the bracketed information in the NUREG-1431 version of Surveillance Requirement 3.6.16.1 is justified. Corresponding changes are also proposed for Surveillance Requirement Bases 3.6.16.1 to reflect the above proposed change. Included in this change is the deletion of the phrase "to the operator," in that there is no direct indication of reactor building door status available in the Control Room to operations personnel at McGuire. During applicable Modes, the opening of these doors is governed through the implementation of strict administrative controls.

Proposed Change 2

TS SR 3.6.16.3 states:

> "Verify reactor building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the reactor building".

The Frequency for this Surveillance is:

> "40 months AND during shutdown for SR 3.6.1.1 Type A tests".

McGuire proposes to amend the Frequency for this Surveillance to state,

> "3 times every 10 years, coinciding with containment visual examinations required by SR 3.6.1.1."

The TS Bases for SR 3.6.16.3 states:

> "This SR would give advance indication of gross deterioration of the concrete structural integrity of the reactor building. The 40 month Frequency is based on the requirement to perform two additional inspections at approximately equal intervals between the Type A tests required by SR 3.6.1.1 performed on a 10-year interval. The verification is done during shutdown."

McGuire proposes to amend the Bases for this Surveillance to state,

> "This SR would give advance indication of gross deterioration of the concrete structural integrity of the reactor building. The Frequency is based on engineering judgment, and is the same as that for containment visual inspections performed in accordance with SR 3.6.1.1."

This change is proposed for the following reasons:

1. The TS Bases SR 3.6.1.1 does not discuss that these two additional inspections be performed "at approximately equal intervals" between the Type A tests, as stated in the TS Bases SR 3.6.16.3.

2. Because Type A tests may be scheduled during a refueling outage that does not coincide with the 40 month frequency for these examinations, a total of 4 examinations could be required during a 10 year interval. This might also occur if McGuire elects to perform Type A tests at a frequency less than 10 years.

3. There are now three separate requirements pertaining to visual examination of the steel containment vessel and the reactor building:

    - 10 CFR 50, Appendix J, Option B, III.A and Regulatory Guide 1.163, Regulatory Position C.3 which requires visual examination of the containment system prior to each Type A test and at 2 other times between Type A tests when the Type A tests are conducted at 10 year intervals.

- 10 CFR 50.55a and the ASME Code, Section XI, Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item E1.11 which require a general visual examination of Class MC components during each ISI Period. Subsection IWE also requires a VT-3 visual examination once each ISI Interval in accordance with Table IWE-2500-1, Examination Category E-A, Item E1.12.

- McGuire TS SR 3.6.16.3 which currently requires a visual examination of the reactor building interior and exterior surfaces prior to each Type A test and every 40 months.

In order to eliminate the performance of unnecessary duplicate examinations to satisfy 10 CFR 50, Appendix J and 10 CFR 50.55a, some flexibility is needed to schedule these general visual examinations within each ISI Period. Because McGuire has historically performed the reactor building examinations in conjunction with the 10 CFR 50, Appendix J containment general visual examinations (and prefers to continue this practice), flexibility in scheduling the reactor building examinations is necessary.

The purpose of the reactor building visual examinations required by TS 3.6.16.3 is to detect deterioration that could affect the structural and leak-tight integrity of the reactor building. The reactor building provides shielding and allows the controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and the Nuclear Steam Supply System. Experience has shown that the structural integrity of the reactor building can be assured by performing these examinations three times every 10 years, including during shutdowns for Type A tests.

Prior to the implementation of the current TS, the reactor building visual examinations were conducted three times every 10 years. Because the proposed change will not reduce the number of examinations required during each 10 year service period, an equivalent level of quality and safety will be provided. The changes to the SR 3.6.16.3 surveillance frequency will allow McGuire to schedule these examinations concurrently with examinations required by 10 CFR 50, Appendix J; the ASME Code, Section XI, Subsection IWE; and 10 CFR 50.55a (b) (2) (ix) (E).

Proposed Change 3

TS Administrative Controls, Section 5.0, 5.5.2 "Containment Leakage Rate Testing Program" states, in part:

> "A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995."

McGuire proposes to amend this requirement by adding the following text:

> "Except that the containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

This change is proposed for the following reason:

> To take exception to Regulatory Position C.3 of Regulatory Guide 1.163 so that visual examinations of the containment performed between Type A tests need not be performed during refueling outages. (Regulatory Position C.3 of Regulatory Guide 1.163 states that these examinations should be conducted during two other "refueling outages".)

The purpose of the containment visual examinations required by 10 CFR 50, Appendix J and Regulatory Guide 1.163, C.3 is to detect deterioration that could affect the containment leak-tight or structural integrity. Performance of these examinations during operation or shutdown has no impact on the quality of the inspection, provided all accessible interior and exterior surfaces are examined. Therefore, McGuire would prefer the option of performing all, or portions, of these visual examinations on accessible interior and exterior surfaces during either plant operation or shutdown, with the exception that examinations performed prior to each Type A test will be performed during the refueling outage.

Note that McGuire intends to perform these general visual examinations concurrently with general visual examinations required by the ASME Code, Section XI, Subsection IWE,

Table IWE-2500-1, Examination Category E-A, Item E1.11 during each ISI Period, as required by 10 CFR 50.55a (b) (2) (ix) (E). The ASME Code does not require that general visual examinations be performed during refueling outages:

Because these visual examinations will be performed during each ISI Period, a minimum of three examinations will be performed every 10 years. Therefore, the revised Technical Specifications provide an equivalent level of quality and safety.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

As required by 10CFR50.91(a)(1), this analysis is provided to demonstrate that the proposed license amendment does not involve a significant hazard.

Conformance of the proposed amendment to the standards for a determination of no significant hazards, as defined in 10CFR50.92, is shown in the following:

1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The proposed amendment to the Technical Specifications does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. Therefore, the proposed amendment does not result in the increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the facility which should introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators, since the containment and reactor building function primarily as accident mitigators.

3) Does the proposed change involve a significant reduction in margin of safety?

   No. Implementation of this amendment would not involve a significant reduction in the margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation, including the performance of the containment and reactor building. The ability of the containment and reactor building to perform their design function will not be impaired by the implementation of this amendment at McGuire Nuclear Station. Consequently, no safety margins will be impacted.

Conclusion

Based on the preceding analysis, it is concluded that the proposed license amendment does not involve a Significant Hazards Consideration Finding as defined in 10CFR50.92.

ATTACHMENT 5

ENVIRONMENTAL ANALYSIS

ENVIRONMENTAL ANALYSIS

The proposed amendment has been reviewed against the criteria of 10CFR51.22 for environmental considerations. The proposed amendment does not involve a significant hazards consideration, increase the types and amounts of effluents that may be released offsite, or result in the increase of individual or cumulative occupational radiation exposures. Therefore, the proposed amendment meets the criteria provided by 10CFR51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.